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NL-16-2480

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555-0001

Joseph M. Farley Nuclear Plant – Unit 1
Licensee Event Report 2016-006-00
Manual Reactor Trip due to Loss of Speed Control
on 1A Steam Generator Feed Pump

Ladies and Gentlemen:

This Licensee Event Report is being submitted pursuant to the requirements of the Code of Federal Regulations, 10 CFR 50.73(a)(2)(iv)(A), for a manual actuation of the Reactor Protection System and an automatic start of the Auxiliary Feedwater system.

This letter contains no NRC commitments. If you have any questions, please contact Julie Collier at 334.814.4639.

Respectfully submitted,

Ms. C. A. Gayheart
Vice President – Farley

CAG/EGA

Enclosure: Unit 1 Licensee Event Report 2016-006-00

cc: Southern Nuclear Operating Company
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Joseph M. Farley Nuclear Plant Unit 1

Unit 1 Licensee Event Report 2016-006-00

Manual Reactor Trip due to Loss of Speed Control on 1A Steam Generator Feed Pump

NRC FORM 366 (11-2015) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 10/31/2018

Estimated burden per response to comply with this mandatory collection request: 80 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the FOIA, Privacy and Information Collections Branch (T-5 F53), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to Infocollects.Resource@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

LICENSEE EVENT REPORT (LER)

1. FACILITY NAME	2. DOCKET NUMBER		3. PAGE
Joseph M. Farley Nuclear Plant, Unit 1	05000	348	1 OF 3

4. TITLE

Manual Reactor Trip due to Loss of Speed Control on 1A Steam Generator Feed Pump

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
11	08	2016	2016	- 006 -	00	12	19	2016	FACILITY NAME	DOCKET NUMBER

9. OPERATING MODE

1

10. POWER LEVEL

32%

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☒ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(C)	☐ 73.77(a)(1)
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(D)	☐ 73.77(a)(2)(i)
☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(vii)	☐ 73.77(a)(2)(ii)
	☐ 50.73(a)(2)(i)(C)	☐ OTHER Specify in Abstract below or in NRC Form 366A	

12. LICENSEE CONTACT FOR THIS LER

LICENSEE CONTACT	TELEPHONE NUMBER *(Include Area Code)*
Julie Collier, Senior Licensing Engineer	334-814-4639

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
B	JK	SC	W120	Y					

14. SUPPLEMENTAL REPORT EXPECTED

☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)* ☒ NO

15. EXPECTED SUBMISSION DATE

MONTH	DAY	YEAR

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

On November 8, 2016, Joseph M. Farley Nuclear Plant Unit 1 was reducing power to remove the main generator from service. The 1A steam generator feed pump did not respond to control steam generator levels as expected when the miniflow valve was opened per procedure. Steam generator levels lowered due to lower feed flow, and at 1331 the reactor was manually tripped from 32 percent power to prevent reaching the low steam generator level automatic reactor trip setpoint. The motor driven auxiliary feed pumps also started automatically, as expected, with the manual reactor trip. The main steam isolation valves were closed to limit the unit cool down, decay heat removal was accomplished with the atmospheric relief valves, and the unit was maintained in mode 3. The controller failure was caused by the speed reference adjust and speed controller (C2) card being out of tolerance due to a failed A2 operational amplifier on the card, which was caused by infant mortality. The C2 card was replaced, and the new card was verified to be within the required tolerance. This event is reportable per 10 CFR 50.73(a)(2)(iv)(A) due to manual actuation of the reactor protection system and automatic actuation of the auxiliary feedwater system.

1. FACILITY NAME	2. DOCKET NUMBER	3. LER NUMBER		
		YEAR	SEQUENTIAL NUMBER	REV NO
Joseph M. Farley Nuclear Plant, Unit 1	05000- 348	2016	006	00

NARRATIVE

A. PLANT AND SYSTEM IDENTIFICATION

Westinghouse - Pressurized Water Reactor

B. DESCRIPTION OF EVENT

At 1326, on November 8, 2016, reactor power was at 34 percent in Mode 1 and power was being reduced to remove the main generator from service. The recirculating flow control valve [FCV] for the 1A Steam Generator Feed Pump (SGFP) [P] was opened in accordance with unit shutdown procedure. All steam generator (SG) [SG] levels began to trend down, as expected and as described in the unit operating procedure. The unit operator attempted to raise speed for the 1A SGFP using the SGFP master speed controller [SC]. The SGFP speed rose to approximately 3700 rpm but would not rise further. The demand on the SGFP master speed controller was greater than 65 percent, which corresponded to a speed greater than 3700 rpm. Due to the inadequate response from the SGFP master speed controller, the Shift Supervisor directed raising speed on the 1A SGFP using the slave controller. Demand on the slave controller was raised to approximately 80 percent, which corresponded to a speed higher than 3700 rpm. The 1A SGFP speed was observed to not have changed and remained at 3700 rpm. The Shift Supervisor then established trip criteria. Due to the inability to raise 1A SGFP speed and feedwater flow to the SGs, all SG levels continued to lower. At 32 percent reactor power, the Shift Supervisor directed a manual reactor trip to prevent reaching the low SG level automatic reactor trip setpoint. At 1331, a manual reactor trip was performed due to lowering SG levels and an inability to raise speed and feedwater flow because the 1A SGFP speed control circuitry failed. The motor driven auxiliary feed pumps also started automatically, as expected, with the manual reactor trip. The main steam isolation valves were closed to limit the unit cool down because the isolation valves [ISV] on the 2A/2B moisture separator reheater [MSR] second stage had excessive leakby. Decay heat removal was accomplished with the atmospheric relief valves, and the unit was maintained in mode 3 to continue with the previously scheduled main generator repairs.

C. UNIT STATUS AT TIME OF EVENT

Unit 1, Mode 1, 32 percent power with a unit shutdown in progress

D. CAUSE OF EVENT

The speed reference adjust and speed controller (C2) card was found to have an out of tolerance output voltage due to an A2 operational amplifier on the card that failed because of infant mortality.

E. REPORTABILITY ANALYSIS AND SAFETY ASSESSMENT

This event is reportable as required by 10 CFR 50. 73(a)(2)(iv)(A) due to a manual actuation of the reactor protection system and automatic actuation of the auxiliary feedwater system. The reactor was shut down at 1331 and mode 3 was entered. There was no loss of safety function

and no radioactive release associated with this event. All required safety systems were available and responded as expected. There were no actual consequences detrimental to the health and safety of the public and the event is considered to be of very low safety significance.

F. CORRECTIVE ACTION

A new C2 card was installed, and the 1A SGFP high pressure and low pressure governor valves were stroked satisfactorily as the functional test.

G. ADDITIONAL INFORMATION

1) Previous Similar Events: none

2) Commitment Information: No commitments are made in this correspondence.